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SOUTHERN COMPANY
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Docket No. 50-366 HL-5531

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Edwin I. Hatch Nuclear Plant - Unit 2
Licensee Event Report
Manual Reactor Shutdown Results in
Water Level Decrease and Group 2 and 5 PCIS Actuations

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv), Southern Nuclear Operating Company is submitting the enclosed Licensee Event Report (LER) concerning a manual reactor shutdown which resulted in a water level decrease and Group 2 and 5 Primary Containment Isolation System (PCIS) actuations.

Sincerely,

Lewis Sumner

H. L. Sumner, Jr.

IFL/eb

Enclosure: LER 50-366/1997-010

cc: *Southern Nuclear Operating Company*
Mr. P. H. Wells, Nuclear Plant General Manager
NORMS

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. L. N. Olshan, Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. B. L. Holbrook, Senior Resident Inspector - Hatch

IE22

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED OMB NO. 3150-0104 EXPIRES: 5/31/95
LICENSEE EVENT REPORT (LER)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Edwin I. Hatch Nuclear Plant - Unit 2	05000366	1 OF 5

TITLE (4)
Manual Reactor Shutdown Results in Water Level Decrease and Group 2 and 5 PCIS Actuations

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER(S)
										05000
11	20	97	97	010	00	12	08	97	FACILITY NAME	05000

OPERATING MODE (9)	1
POWER LEVEL (10)	071

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR ? : (Check one or more of the following) (11)

20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (include area code)
Steven B. Tipps, Nuclear Safety and Compliance Manager, Hatch	AREA CODE 912 367-7851

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	SD	None	C601	Yes					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-space typewritten lines) (16)

On 11/20/97 at 2053 EST, Unit 2 was in the Run mode at a power level of 1816 CMWT (71 percent rated thermal power). At that time, personnel manually scrammed the reactor so the condensate system could be isolated to repair damage. Water level decreased due to void collapse from the rapid reduction in power resulting in Group 2 and 5 primary containment isolation system isolation signals and closure of the Group 2 and 5 primary containment isolation valves. Level reached a minimum of approximately 38 inches below instrument zero (120.44 inches above the top of the active fuel) causing automatic initiation of the reactor core isolation cooling (RCIC) and high pressure coolant injection systems. Level was recovered by the reactor feedwater pumps (RFPs) before either system could inject to the vessel. Secondary containment automatically isolated and the Unit 1 and Unit 2 standby gas treatment systems automatically started per design. Level was restored to normal within one minute by the RFPs and maintained by the RCIC system. Pressure reached a maximum value of 990 psig; no safety/relief valves lifted nor were any required to lift.

This event was caused by component failure. The hinge pin in discharge check valve 2N21-F019A broke resulting in the valve failing to close when condensate booster pump 2N21-C002A was removed from service. This caused high pressure water to damage the pump and its low pressure suction piping and bellows. Corrective actions include repairing the damage. The failure of the hinge pin appears to be an isolated case.

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes appear in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

On 11/20/97 at 2053 EST, Unit 2 was in the Run mode at a power level of 1816 CMWT (71 percent rated thermal power). At that time, Operations personnel manually scrammed the reactor so the condensate system could be isolated to remove pressure from and repair damage to condensate booster pump 2N21-C002A (EIIS Code SD) and its suction piping. The pump and suction piping had been damaged at 1430 EST on 11/20/97 when condensate booster pump discharge check valve 2N21-F019A (EIIS Code SD) failed in the open position after pump 2N21-C002A had been removed from service for maintenance. The failure of discharge check valve 2N21-F019A to close allowed high pressure condensate from the discharge of the operating condensate booster pumps to flow through the minimum flow line for pump 2N21-C002A, the booster pump, and its low pressure suction line to the suction header for the three booster pumps. The back flow through the pump and suction line caused pump 2N21-C002A to spin backwards and exposed the suction line expansion bellows to pressures above its rating of 150 psig and hydrostatic test pressure of 300 psig. The sudden pressure perturbation also dislocated the suction piping at the expansion bellows by approximately three inches causing additional damage to the bellows.

Following confirmation from Hatch's Architect/Engineer that the suction line expansion bellows was not rated for high pressures and could not be isolated, management ordered the unit shut down. At 2034 EST, Operations personnel began reducing reactor power by decreasing reactor recirculation pump (EIIS Code AD) flow rate. They also began to open the condensate booster pump minimum flow valve per procedure 34SO-N21-007-2S, "Condensate and Feedwater System," to lessen the pressure increase on the damaged suction line bellows as reactor power was decreased. At 2052 EST, the power decrease was halted at 71 percent rated thermal power to avoid entering the region of potential instabilities on the power-to-core-flow map. Also, the minimum flow valve was fully open and the power reduction was causing pressure on the damaged suction bellows to increase to approximately 475 psig. Operations personnel then increased reactor vessel water level approximately five inches to 40 inches above instrument zero in anticipation of the resulting level decrease from a scram at high power and in recognition of the increased potential for loss of the reactor feedwater pumps (EIIS Code SJ) from the damage to the condensate system. At 2053 EST, Operations personnel manually scrammed the unit.

Subsequent to the manual scram, vessel water level decreased due to void collapse from the rapid reduction in power. The decrease in water level resulted in receipt of Group 2 and Group 5 primary containment isolation system (EIIS Code JM) isolation signals and closure of the Group 2 and Group 5 primary containment isolation valves (EIIS Code JM) per design. Water level reached a minimum of approximately 38 inches below instrument zero (120.44 inches above the top of the active fuel). The level decrease resulted in automatic initiation of the reactor core isolation cooling (RCIC, EIIS Code BN) and high pressure coolant injection (HPCI, EIIS Code BJ) systems on low low reactor water level per design. However, water level was recovered by the operating reactor feedwater pumps before either system could inject to the reactor vessel. Operations personnel secured the HPCI and RCIC systems with water level at three inches above instrument zero and increasing.

Secondary containment automatically isolated and the Unit 1 and Unit 2 standby gas treatment systems (EIIS Code BH) automatically started on low low reactor water level per design. Water level was restored to normal within one minute of the manual scram by the reactor feedwater pumps. Operations personnel manually tripped the "B" reactor feedwater pump; the "A" reactor feedwater pump subsequently tripped on low suction pressure. Operations personnel did not attempt to restart a reactor feedwater pump because pressure perturbations resulting from the manual scram caused additional damage to the condensate system suction piping and expansion bellows. Reactor vessel water level was maintained by manual operation of the RCIC system.

Reactor vessel pressure reached a maximum value of 990 psig. No safety/relief valves lifted nor were any required to lift to reduce or control pressure. Reactor vessel pressure was controlled initially using the main turbine bypass valves (EIIS Code SO). Following the opening of the main condenser vacuum breakers (EIIS Code SH) at 2323 EST, Operations personnel used the HPCI and RCIC systems for pressure control.

CAUSE OF EVENT

This event was caused by component failure. The hinge pin in condensate booster pump discharge check valve 2N21-F019A broke resulting in the valve failing to close when pump 2N21-C002A was removed from service at 1430 EST on 11/20/97. The failure of the discharge check valve to close caused high pressure water to flow through pump 2N21-C002A into its suction line damaging the pump and the low pressure suction piping and bellows.

The reason the hinge pin broke is not known; however, a review of the maintenance history for the Unit 2 condensate booster pump check valves since 1986 reveals no previous instances of broken hinge pins or of a valve failing to close for any reason. Therefore, it is reasonable to conclude that this event is an isolated occurrence.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73 (a)(2)(iv) because of the unplanned actuation of an Engineered Safety Feature system. Following insertion of a manual reactor scram, water level decreased due to void collapse. The decrease in water level resulted in the receipt of automatic Group 2 and Group 5 primary containment isolation system isolation signals on low and low low reactor water level, respectively, and closure of the Group 2 and Group 5 primary containment isolation valves per design. The primary containment isolation system is an Engineered Safety Feature system.

The condensate (EIIS Code SD) and feedwater (EIIS Code SJ) system supplies feedwater to the reactor vessel. The condensate pumps (EIIS Code SD) take condensate from the main condenser (EIIS Code SQ) hotwells and pump it through the air ejector condensers (EIIS Code SH), the gland seal condenser, and the condensate demineralizers (EIIS Code SF). The condensate booster pumps take the demineralizer effluent and pump it through two parallel streams of four low pressure heaters (EIIS Code SD) to the suction of the reactor feedwater pumps. The reactor feedwater pumps then pump the feedwater through two parallel streams of one high pressure heater (EIIS Code SJ) to the reactor vessel.

In this event, Operations personnel manually scrammed the reactor so the condensate system could be isolated to remove pressure from and repair damage to condensate booster pump 2N21-C002A and its suction piping. The pump and suction piping had been damaged when condensate booster pump discharge check valve 2N21-F019A failed in the open position after the pump had been removed from service for maintenance. Following the manual reactor scram, water level decreased due to void collapse from the rapid reduction in power. The decrease in water level resulted in receipt of Group 2 and Group 5 primary containment isolation system isolation signals and closure of the Group 2 and Group 5 primary containment isolation valves per design. Level reached a minimum of about 38 inches below instrument zero (120.44 inches above the top of the active fuel) resulting in automatic initiation of the RCIC and HPCI systems per design. However, water level was recovered by the operating reactor feedwater pumps before either system could inject to the reactor vessel. Level was restored to normal within one minute by the reactor feedwater pumps and maintained by the RCIC system. Secondary containment automatically isolated and the Unit 1 and Unit 2 standby gas treatment systems automatically started on low water level per design. Pressure reached a maximum value of 990 psig; therefore, no safety/relief valves lifted nor were any required to lift to reduce or control pressure.

All systems functioned as expected and per their design given the water level and pressure transients. Water level was maintained well above the top of the active fuel throughout the transient and was restored to normal within one minute of the manual scram. Therefore, it is concluded the event had no adverse impact on nuclear safety. This analysis is applicable to all power levels.

CORRECTIVE ACTIONS

Discharge check valve 2N21-F019A was removed, inspected, and rebuilt with new plates, hinge pin, spring, and bearings. In addition, the internals of condensate booster pump discharge check valve 2N21-F019B were inspected. Maintenance personnel found the spring broken and a small amount of wear on the hinge pin; they replaced the spring and the hinge pin. Condensate booster pump discharge check valve 2N21-F019C was not inspected because it had been inspected and repaired in March 1997. The failure of the hinge pin appears to be an isolated case.

The damaged suction piping and bellows were repaired. Condensate booster pump 2N21-C002A will be inspected and repaired/rebuilt as conditions and manpower allow.

ADDITIONAL INFORMATION

No systems other than those already mentioned in this report were affected by this event.

Failed Component Information:

Master Parts List Number: 2N21-F019A
Manufacturer: Control Equipment
Part Number: 10304-289-52
Type: Hinge Pin
Manufacturer Code: C601

EIIS System Code: SD
Reportable to NPRDS: Yes
Root Cause Code: X
EIIS Component Code: (None)

No previous similar events have occurred in the last two years in which a failed check valve has directly or indirectly resulted in an unplanned Engineered Safety Feature system actuation.